NON-PUBLIC?: N
ACCESSION #: 8805020281

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 1 PAGE: 1 of 12

DOCKET NUMBER: 05000220

TITLE: Manual Reactor Scram Initiated Due To Feedwater Piping Vibration
EVENT DATE: 12/19/87 LER #: 87-028-01 REPORT DATE: 04/21/88

OPERATING MODE: N POWER LEVEL: 098

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Peter A. Mazzaferro, Assistant Supervisor, Technical Support
TELEPHONE #: 315-349-2190

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SJ COMPONENT: FCV MANUFACTURER: F130
REPORTABLE TO NPRDS: Y
CAUSE: B SYSTEM: EC COMPONENT: CL MANUFACTURER: E020
REPORTABLE TO NPRDS: N
CAUSE: X SYSTEM: SJ COMPONENT: P MANUFACTURER: W318
REPORTABLE TO NPRDS: Y
CAUSE: X SYSTEM: SJ COMPONENT: SNB MANUFACTURER: I207
REPORTABLE TO NPRDS: Y
CAUSE: X SYSTEM: SJ COMPONENT: P MANUFACTURER: W318
REPORTABLE TO NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT: At 18:13:56 on December 19, 1987, Nine Mile Point Unit 1 (NMP1) was manually scrammed due to excessive vibration observed in the high pressure feedwater piping. The root cause of this event was determined to be fatigue fracture of the stem/valve plug assembly of Flow Control Valve 13A due to flow-induced vibration. The valve failure caused pressure surges which resulted in vibration of the feedwater piping and damage to adjacent pipe supports. The pressure-retaining boundary of the feedwater system was not damaged by this transient. A detailed visual inspection of the feedwater system piping and supports was conducted. Non-destructive examination of selected pipe welds in the feedwater system which were subjected to the

highest stresses during the transient was performed with no discrepancies noted. The stem/valve plug assembly has been replaced with a new design which should provide greater reliability.

This supplement is being submitted to describe the final determination of the root cause, the results of the inspections and analyses completed, and the corrective action taken. An additional supplement is not expected to be submitted.

(End of Abstract)
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COMPONENT FAILURE DESCRIPTION #5 MOVED TO PAGE 1 OF 1

TEXT: PAGE: 3 of 12

I. DESCRIPTION OF THE EVENT

At 18:13:56 on December 19, 1987, Nine Mile Point Unit 1 (NMP1) was manually scrammed due to excessive vibration observed in the high pressure feedwater piping.

Prior to the event, the unit was on-line with the reactor operating at approximately 98% of rated full power. Feedwater Pump (FWP) #13, driven through a clutch/gear arrangement by the main turbine, was in operation and supplying approximately 80% of the total feedwater flow to the reactor. The pump utilizes dual air-operated flow control valves (FCV) (designated 13A and 13B) arranged in parallel, with both valves positioned by a single control signal from the Feedwater Control System. FWP #11 (motor-driven) was also in service, with its single air-operated FCV manually adjusted from the Control Room to provide the balance of the feedwater flow. FWP #12 (also motor-driven) was off, in the standby condition. Reactor water level was stable at approximately 78 inches. At approximately 1810, a fire detector located inside the Turbine Lube Oil Reservoir Room tripped to alarm, and the duty Fire Chief was notified by the Control Room to investigate (the alarm wa later identified as being caused by excessive dust in the air released by movement of the feedwater piping). At approximately the same time, random high and high-high water level alarms on the Feedwater Heaters were received in the Control Room. These alarms were predominately on the fourth stage low pressure and the high pressure Feedwater Heaters (Post-event evaluation indicates that most of these alarms were probably caused by the level instrumentation on the Feedwater Heaters vibrating, and not actual measured high water level). While investigating the cause of these alarms, the Chief Shift Operator (CSO) noted an indication on a strip chart recorder that the total feedwater flow to the reactor was flucuating slightly. He immediately took remote-manual control of FCV 13A and 13B (operated together from a single controller) in an attempt to eliminate the feedwater instability. Feedwater flow continued to respond erratically, and an operator was sent into the Turbine Building to observe the stroking of the valve stems on FCV's 13A and 13B. Control problems with FCV 13A and 13B had been encountered on December 7, 1987, which resulted in an automatic reactor low-level scram (reported in LER 87-24). With this in mind, the CSO began to reduce feedwater flow with FCV 13A and 13B, while another licensed operator initiated a reactor power reduction with recirculation flow at approximately 18:11:45. The intent was to reduce reactor power below the point where sufficient feedwater flow could be supplied by #11 and #12 FWP's, and then secure #13 FWP. At 18:12:00, a #13 FWP trouble alarm was received in the Control Room. It cleared immediately and then began to alarm intermittently. FWP #12 was started at 18:13:15 in preparation for securing #13 FWP as soon as possible and to provide additional feedwater capacity should it be needed. The alarms on #13 FWP and the Feedwater Heater levels continued intermittently throughout the power reduction. The operator dispatched to observe FCV 13A and 13B reported that the floor in the vicinity of #13 FWP seemed to be vibrating more than usual and that he could not view the stem motion on the FCV's because both the valves and the feedwater piping were shaking violently. At approximately 18:13:50, the operators inside the Control Room sensed vibrations in the

floor. The Station Shift Supervisor (SSS) ordered a manual reactor scram, which was initiated at 18:13:56. Reactor power had been reduced to approximately 79% at the time of the scram. Immediately following the scram, the CSO tripped the clutch on #13 FWP, believing that it could be the cause of the vibration. The vibrations felt in the Control Room lasted a total of about 5 seconds and ceased at approximately the same time as the reactor scram and the #13 FWP clutch disengagement. A second fire alarm was received at approximately 18:15 in the Condenser Bay area immediately adjacent to the previously alarming zone (same cause as the first alarm).

Reactor water level dropped to 53 inches 5 seconds after the manual scram. This caused the feedwater system to shift into the High Pressure Coolant Injection (HPCI) mode of operation and initiated an automatic reactor scram signal. The automatic reactor scram caused a sequential trip of the main turbine and the generator tripped on reverse-power as designed. Reactor water level dropped to approximately 19 inches before recovering, then overshot the normal control band. Both motor-driven FWP's tripped on high reactor water level of 95 inches at 18:16. The operators then proceeded to place the unit in a stable shutdown condition without further incident.

Post-trip walkdown of the feedwater system showed evidence of pipe motion during the transient extending from the suction side of #13 FWP to the high pressure feedwater heaters. Insulation was damaged where the piping passed through wall or floor sleeves. The pressure-retaining boundary of the piping system was not damaged by this transient. The cause of the fire alarms was identified as dust released by the damaged pipe insulation tripping the photoelectric fire sensors. Both detectors remained in the alarming condition for about 10 minutes before clearing.

II. CAUSE OF THE EVENT

The root cause of this event was determined to be flow-induced vibration of the plug in FCV 13A. This vibration induced a fatigue fracture in the valve stem, allowing the plug to oscillate open-close and establish damaging vibrations in the feedwater piping. Increased stress concentration at a small-radius fillet weld (connecting the stem to the plug) is believed to be a contributing factor.

Disassembly and inspection of the internals of FCV 13A (Atta
hment 1)
revealed that the valve plug had separated from the valve stem. This was a result of a fracture through the weld between the stem and the plug which seals the threaded connection between the 1-inch diameter shaft and the valve plug (Attachment 2). In addition, there were marks on both the valve plug and on the valve seat cages, indicating that the plug had been hammering against the webbing between the ports in the cages. The plug in those areas had been worn such that indentations several mils deep had been made in the plug circumference. The failed valve plug and stem were sent offsite for metallographic analysis to help determine the cause and mechanism of

failure. Visual examination of the valve body was performed to determine if the event had damaged the body. No indication of damage was found. Inspection of valve plug/stem assembly of FCV 13B, which was of an identical design to 13A, revealed no sign of unusual wear or damage. The weld on the FCV 13B valve plug/stem had a significantly larger radius (9/16-inch) than that on the corresponding weld on FCV 13A. There was evidence of hammering or wear between the valve and the adjacent cages, but not to as severe an extent as on FCv 13A.

Metallurgical analysis of the FCV 13A valve plug and stem indicated that the weld failure was a reverse bending fatigue fracture. The bending stresses were believed to have developed in service as a result of wear on the lower plug. Flow-induced vibration over a long period of time wore the contact surface between the plug and the cage and, through increased amplitude of vibration caused by this wear, caused the stem to fail in fatigue.

Hydraulic analysis shows that when the valve stem fracture occurred, the plug was suddenly driven closed. A pressure difference was set up across the valve which propagated through the piping system, with a compression wave upstream of the valve and a decompression wave downstream of the valve. A direct force was applied to the valve as a result of this pressure difference, and forces were also applied at each elbow in the piping system as the pressure wave reached those elbows. Flow and pressure conditions in the valve were such that the separated plug was caused to open when it was closed and to close when it was open. For any force applied to the plug by the fluid, an equal but opposite force was applied to the valve body. Since the piping system is not highly restrained vertically near the feedwater control valves, the reaction force on the valve added to the vertical motion of the pipe near the valve and accentuated the effect of the pressure pulsations in moving the pipe, thus affecting the relative position of the plug in the valve. The most likely effect is that the plug motion tunes itself to the natural frequency of the pipe and that fluid forces caused by the relative motion of the plug and the valve will act to reinforce that motion. This motion caused the piping system near the valve to vibrate at its natural frequency, thereby causing sufficient amplitude of the vibrations to damage the pipe supports.

III. ANALYSIS OF THE EVENT

There were no adverse safety consequences associated with this event nor was the reactor in an unsafe condition at any time. The high vibration experienced in the feedwater system was terminated without adverse consequences to any system required to shutdown the reactor. All operator-initiated and automatic safety systems operated as designed.

III. ANALYSIS OF THE EVENT (Cont'd)

With respect to the potential safety consequences of the worst case scenario of a feedwater pipe break, the following evaluation is presented. Static load/displacent and dynamic load analyses of the high pressure feedwater system were conducted to analytically simulate the observed motions of the piping system and to locate the higher stress locations in the system. The piping reducers adjacent to the FCVs were identified as the highest stress point in the dynamic piping model. Field inspections of the feedwater system piping and supports revealed no evidence of pipe vibration inside of primary containment. Therefore, the event can be evaluated from the perspective of a postulated failure of a high energy piping system outside of primary containment. This event was analyzed in the NMP1 Final Safety

Analysis Report (Updated), Section XVI, 2.0, "Plant Design for Protection Against Postulated Piping Failures in High Energy Lines." Subsection 2.2 analyzed the failure of each high energy piping system at any point outside of primary containment and concluded that safe shutdown of the reactor can be accomplished and the unit could be maintained in the shutdown condition. In addition, a more detailed analysis of a high pressure feedwater pipe rupture also concluded that a failure at any point outside of primary containment would not affect the safe shutdown capability of the reactor. Therefore, the potential safety consequences of this feedwater transient would not have placed the reactor in a condition not previously analyzed.

IV. CORRECTIVE ACTION

An extensive visual examination of the feedwater system and its associated pipe supports and shock suppressors was conducted. Inspections were carried out on the 322 feedwater system piping supports. Some 84% of the supports were found to be unaffected; 10% were found to need minor repair action, with the need for repair attributed to normal operation and not the transient; and 6% (18 of the 322) were the subject of Nonconformance Reports and required repair or rework (these were assumed to have been damaged in the transient). Observations were made on spring hanger settings/locations and pin-to-pin dimensions on snubbers to determine whether any permanent deformation of the feedwater piping had occurred as a result of the transient. The conclusion drawn from this evaluation was that no permanent deformation of the piping had resulted.

Static and dynamic analyses of the high-pressure feedwater system piping were performed by the Nuclear Engineering Department using SUPERPIPE, an industry recognized computer program. This part of the feedwater system was selected for analysis based upon the obvious physical damage observed on the initial walkdown and the operators observation of significant vibration in the area. The balance of the feedwater system was not analyzed, as the high-pressure section was judged to have experienced the most severe effects. These analyses concluded that: (1) the observed damage at piping supports can be explained by the proposed hydraulic scenario and the piping dynamic analyses, (2) the feedwater system was not overstressed (beyond ASME Boiler

IV. CORRECTIVE ACTION (Cont'd)

and Pressure Vessel Code, Section III) due to the hydraulic transient, and (3) the high-pressure feedwater system piping integrity has not been compromised by this transient event and is safe for plant start-up and continued operation. The analyses also indicated that the highest stressed portion of the piping in this system was in the reducers adjacent to the FCVs.

Surface examination was performed on 22 welds on the high-pressure feedwater system piping. The welds selected were those which were determined by analysis to have been subjected to the highest stresses during the feedwater transient. Of the 22 welds examined, 2 had rejectable indications. However, these indications were attributed to original imperfections that were acceptable to the original material specification and not to the effects of the feedwater transient. The rejectable indications were all removed by grinding or flapping and the surface accepted for service.

Functional testing was performed on 5 hydraulic shock suppressors on the high-pressure feedwater piping that calculations indicated had been subjected to large loads during the feedwater transient. They were selected on the basis of the piping stress analysis performed by Nuclear Engineering. Shock suppressor 29-HS-11, which had been visibly bent during the transient, was also tested to determine whether it had functioned properly during the transient. It was determined that all of the tested shock suppressors were functional and had performed as designed during the transient. All of the shock suppressors tested, with the exception of 29-HS-11, were returned to service. 29-HS-11 was replaced with a new shock suppressor.

An in-service leak test will be performed on all feedwater system equipment which had been opened for inspection or repair during the system startup.

NMP1 has experienced problems in the past with the stem separating from the valve plug on these particular FCV's because of the severe operating environment. The original valve plug/stem design, which featured a 1-inch diameter shaft screwed into the plug and pinned with a 1/4-inch diameter pin, was changed to the present design in 1979. Since then the valve internals have been replaced at every 2-year refueling cycle interval. FCV #11 and #12 were also disassembled and checked for damage. These valves have not had a history of stem failures as experienced by 13A and 13B and no damage was noted during the inspection.

A redesigned valve plug/stem assembly has been installed in FCV 13A and 13B (Attachment 2). The attachment point has been strengthened by increasing the diameter of the stem where it attaches to the plug and adding a conical section to reduce the stress intensification at the attachment point. A 3/8-inch diameter pin is used to secure the valve plug to the stem.

V. ADDITIONAL INFORMATION

FWP #11 auxiliary oil pump did not start after the pump had tripped on high reactor water level during the event. A work request was generated to investigate this and to inspect the pump bearings for any damage. While in the process of removing the insulation from the pump casing, Maintenance personnel noticed 4 distinct pinhole leaks in the factory weld between the carbon steel weld and the 5% chrome pump casting. Because of the minute size and configuration of the holes, there was no measurable leakage rate from the affected area. NDE was performed on the pump casing, suction, and discharge nozzles with no other leakage paths identified. A through-wall crack was discovered in the area of the pinholes and two additional indications of possible flaws in this weld were also observed.

A boat sample was taken from the weld in the area of the pinholes and examined to determine the most probable cause of failure. The results of the investigation indicated that the crack most likely was caused by intergranular stress corrosion cracking (IGSCC). The weld material was found to be sensitized and, thereby, was susceptable to IGSCC. The stresses which assisted intergranular corrosion were believed to be a combination of residual welding stresses and applied service stresses. The weld was a factory weld, made by the pump vendor, and signs of the initial weld repair were evident.

The suction elbow was examined as part of the erosion/corrosion program. All areas had sufficient minimum wall thickness. Several areas, however, did not have the full corrosion allowance. Consequently, the wall thickness will be monitored at subsequent outages. Weld repairs were made where required. The pump will be hydrostatically leak-tested before being returned to service. NDE was also performed on #12 FWP with no discrepancies noted. The failure of #11 FWP auxiliary oil pump was traced to a defective coil in the timer motor control circuit. The failure of this coil was not related to the feedwater transient.

FWP #13 was disassembled following the event to inspect the pump bearings. During the transient, intermittent low lube oil pressure, seal water differential pressure, and inboard seal level alarms were received. Inspection of the pump internals revealed that the outboard line bearing top half was galled and that the inboard seal and remainder of the outboard seal parts showed evidence of rubbing. A triangular-shaped piece of metal about 2 square inches in area had broken off of one of the six inlet turning vanes on the impeller and was missing. Inspections were made in the pump casing, the accessible portions of the pump suction and discharge piping, and the valve bodies of FCV 13A and 13B in an attempt to locate the missing piece. The internals of the flow check valve immediately upstream of the FCV's was also inspected. The damaged impeller of the FWP #13 was examined to determine the cause of failure. As the broken piece could not be found, the examination covered only the impeller. It was concluded that a casting defect in the affected inlet turning vane was the primary cause of failure. The defect in

V. ADDITIONAL INFORMATION (Cont'd)

the vane was activated by the removal of surface material by cavitation erosion. A crack resulted, and the pressure pulse created by the closing of FCV 13A (when its stem failed) was of sufficient force to cause the missing piece to break away from the vane. A loose parts analysis was performed to determine the possible safety consequences the lost part could create. The report concluded that the introduction of the missing piece from the FWP #13 impeller would not adversely affect the operation of any component in the feedwater system or the reactor vessel so as to constitute an unreviewed safety question. The normal preventive maintenance for FWP #13 calls for the internals of this pump (the volute and the impeller) to be replaced at the scheduled refueling outage every two years. This was accomplished, as the parts were available, awaiting the refueling outage scheduled to begin in March 1988.

The following is a list of all Nonconformance Reports (NCR) generated as a result of this transient. Some of the items discovered during the various inspections and examinations appear not to be related to this transient and probably were already existing when the event occurred.

NCR On Items
NCR Attributed To Not Resulting From
Feedwater Transient Feedwater Transient

1-87-0077 (Snubber) 1-87-0080 (FWP #11)
1-87-0078 (Snubber) 1-87-0087 (FWP #11)
1-87-0081 (Snubber) 1-87-0088 (Support)
1-87-0082 (Snubber) 1-87-0089 (FWP #11)
1-87-0083 (Snubber) 1-87-0090 (FWP #11)
1-87-0086 (Supports and Snubbers) 1-88-0002 (Support)
1-87-0091 (Support) 1-88-0005 (Weld)
1-87-0001 (Support) 1-88-0004 (Weld)
1-88-0003 (Support) 1-88-0007 (Weld)
1-88-0006 (Snubber) 1-88-0008 (Support)
1-87-0079 (Pipe) 1-88-0009 (Weld)
1-88-0017 (Bolt)
1-88-0018 (Bolt)
1-88-0019 (Bolt)
1-88-0020 (Bolt)
1-88-0021 (Bolt)

1-88-0022 (Bolt)
1-88-0024 (Support)

V. ADDITIONAL INFORMATION (Cont'd)

A search of NMP1 records indicates that a similar feedwater transient, resulting in a manual reactor scram, occurred on January 20, 1978. This event was not reported in an LER due to different reporting criteria in effect at that time. The unit was initially on-line, with the reactor operating at full power. Reactor water level control became unstable due to an apparent problem with FCV 13A. A power reduction was initiated and attempts were made to take local control of the FCV. The SSS ordered a manual reactor scram initiated while at 78% power due to violent vibration of the feedwater piping. Post event examination of the internals of FCV 13A revealed that the stem had separated from the valve plug. The set pin which locked the threaded valve stem to the plug was found to be sheared, and the valve plug became unscrewed from the stem. During the subsequent unit startup, FCV 13B suffered a fractured stem and the plug was no longer attached to the stem. As a result of these failures, the design was modified in 1978 and installed in 1979 as previously described in Section IV.

Identification of Components referred to in this LER.

Component NPRDS
Component Model Code Vendor/Mfgr. Vendor Code

Flow Control 476L-5-HSV FCV Fisher Controls F130
Valve 13A Size 86, Co.
10-inch

Feedwater 8WNC141 P Worthington Pump W318
Pump #11 Corp.

Feedwater 18WNC191 P Worthington Pump W318
Pump #13 Corp.

Coil HA-25 CL Eagle Signal E020

Shock Fig. 201 SNB ITT Grinnell I207
Suppressor

ATTACHMENT 1

SKETCH OF THE FLOW-CONTROL VALVE SHOWING THE STEM/PLUG COMPONENT

FIGURE OMITTED - NOT KEYABLE (DRAWING)

ATTACHMENT 2

FIGURE OMITTED - NOT KEYABLE (DRAWING)

NIAGARA NMP 33504
MOHAWK

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April 21, 1988

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-220
LER 87-28, Supplement 1

Gentlemen:

In accordance with 10 CFR 50.73, we hereby submit the following Licensee Event Report:

LER 87-28, Which is being submitted in accordance with 10 CFR 50.73
Supplement 1 (a)(2)(iv), "Any event or condition that resulted in manual
or automatic actuation of any Engineered Safety Feature
(ESF), including the Reactor Protection System
(RPS). However, actuation of an ESF, including the RPS,
that resulted from and was part of the preplanned sequence
during testing or reactor operation need not be reported."

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September, 1985.

Very truly yours,
/s/ Thomas J. Perkins
Thomas J. Perkins
Vice President, Nuclear

TJP/meh

Attachment

cc: William T. Russell
Regional Administrator

*** END OF DOCUMENT ***